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Thermal Stress Experiments

On NERVA Fuel Material



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I. Introduction

NERVA reactor fuel material operates at relatively high power densities of 2-4 Kw/cm³. Under normal operation this level of heat generation leads to relatively modest stress levels of the order of 500 psi. However, abnormal operating conditions may be caused by plugged coolant channels or hydrogen in-leakage at the core periphery. Both of these conditions would serve to increase local temperature differences and, thus, local stress levels. Under these conditions stress levels up to 2000 psi have been estimated. In order to investigate both normal and abnormal core behavior, thermal stress experiments were performed by pulsed nuclear heating in the transient reactor test facility (TREAT). Two types of nuclear pulses were utilized to mockup, respectively normal and abnormal core thermal conditions, namely slow or extended nuclear pulses and fast nuclear pulses.

II. Purpose of Experiments

The purpose of these experiments was three fold:

- (1) To evaluate the thermal stress behavior of NERVA reference fuel at power densities and temperature differences comparable to those experienced in the NERVA reactor during normal operation.
- (2) To evaluate the thermal stress behavior of full-cross section (3/4 inch, 19 hole) NERVA fuel material subjected to large temperature differences; i. e., at power densities in excess of those anticipated during normal NERVA reactor operation.
- (3) To compare the behavior of cylindrical (1/2 inch diameter, 7 hole) fuel material cut from the center of NERVA reference fuel to the behavior of 19 hole reference fuel and to the behavior of fuel material previously tested at Argonne National Laboratory.* (The work on early experimental KIWI fuel material manufactured at LASL had indicated a relatively low threshold for thermal stress cracking of the inside cylindrical material cut from a hexagonal shape.)

* Vogel, M. A., WANL-TNR-139, Irradiation Studies on NERVA Fuel Material - CY 1963.

III. Experimental Techniques, Equipment, and Fuel Material Tested

A. Experimental Techniques

Transient irradiations were carried out in Argonne National Laboratory's TREAT reactor at the National Reactor Testing station in Idaho. TREAT is a homogeneous, graphite-moderated and graphite-reflected, transient nuclear reactor. The core is composed of graphite-urania matrix fuel elements encased in zircaloy cans. The overall dimensions of each fuel element including reflector graphite are 4 in. x 4 in. x 8 ft. The reactor fuel has a carbon to uranium-235 ratio of 10,000:1. The large amount of carbon serves as moderator, as a large heat sink and provides a negative temperature coefficient of reactivity. The reactor is capable of developing transient thermal flux integrals of approximately 3.5×10^{15} nvt thermal. The maximum energy release of the reactor is 1000 Mw-sec with a period of 40 milliseconds. (See Figure 1)

The experimental procedure used for each transient pulse consists of bringing the reactor to a nominal steady-state power level of 10 w. Excess reactivity is then added above prompt critical to initiate the pulse. The total energy of the pulse is temperature limited by the negative temperature coefficient of the TREAT fueled graphite elements, or the pulse can be clipped by rapidly inserting control rods. The period and peak power of a pulse can be controlled by the addition of the proper excess reactivity. Thus, it is possible to exercise control over both the rate of energy input (period) and the total energy input (integrated power). Fuel sample heatup during fast pulses (less than 75 ms period) is essentially an adiabatic process. Any desired fuel sample temperature (temperatures greater than 3500°C have been attained) may be achieved by selecting the proper value of integrated power for a given size fuel sample and for a particular capsule arrangement. The extended or "flat-top" nuclear pulses are carried out in a similar manner except that before the negative temperature coefficient of reactivity begins to shut down the reactor, positive reactivity is added to sustain the reaction by removing a control rod at a predetermined rate. In this manner relatively constant temperature pulses of 10-20 seconds can be attained.

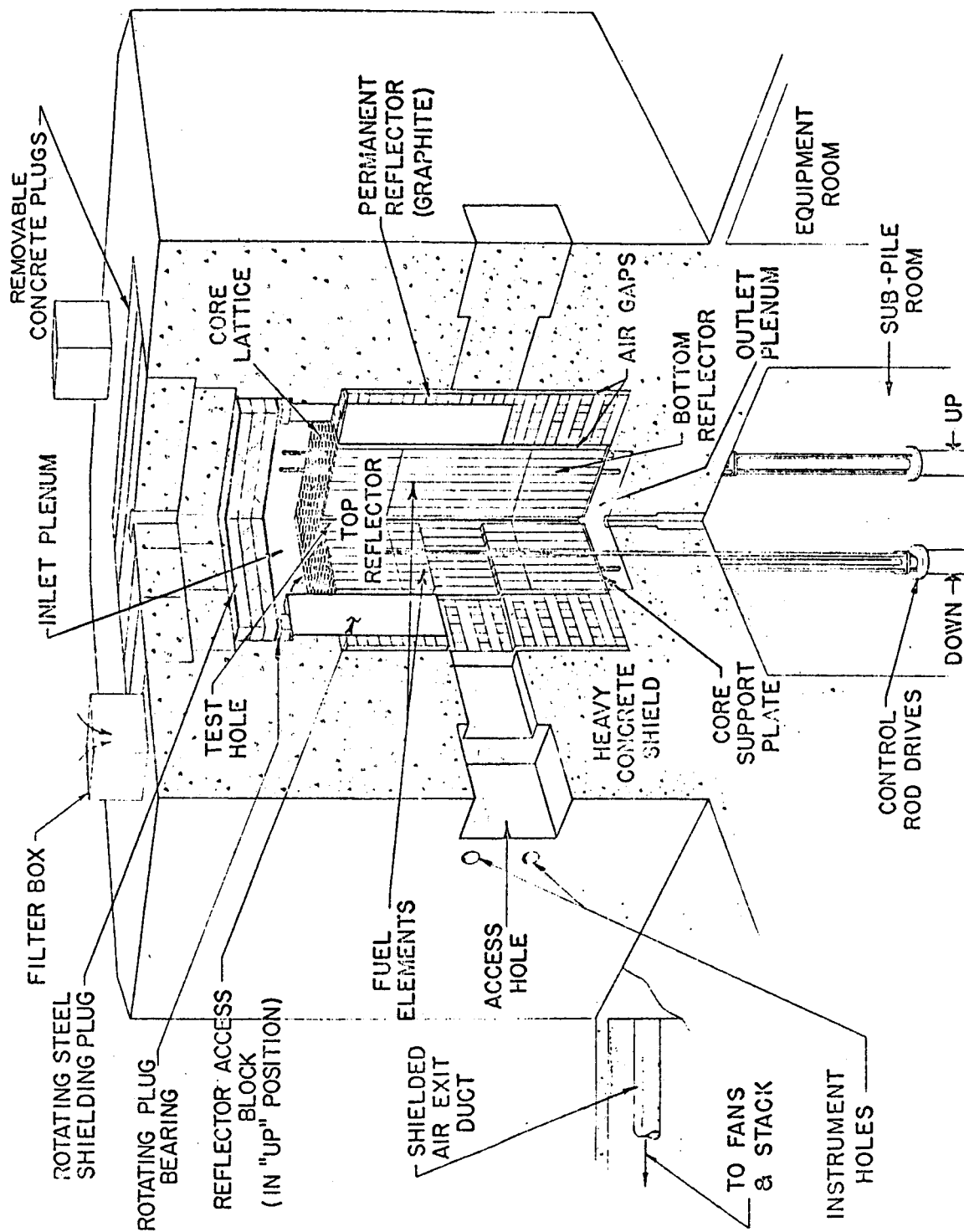


Figure 1 - Perspective View of TREAT

B. Equipment

Figure 2 shows the design of the experimental assemblies which were used for the four transient irradiation experiments. The assemblies consisted of stainless steel autoclave vessels designed for horizontal insertion into the TREAT slot facility. The autoclaves had an internal volume of 3600 cm³ and were designed for a maximum operating pressure of 2000 psi at 800°F. Internal hardware (Figure 3) consisted of three concentric graphite crucibles with top* and bottom plugs to contain the 6-inch long hexagonal test sample. The bottom graphite plugs on capsules A-2 and A-3 were provided with a cavity to contain 1/2 inch diameter, 3/4 inch long fuel samples. Thus, in these experiments cylindrical samples were tested along with hexagonal samples.

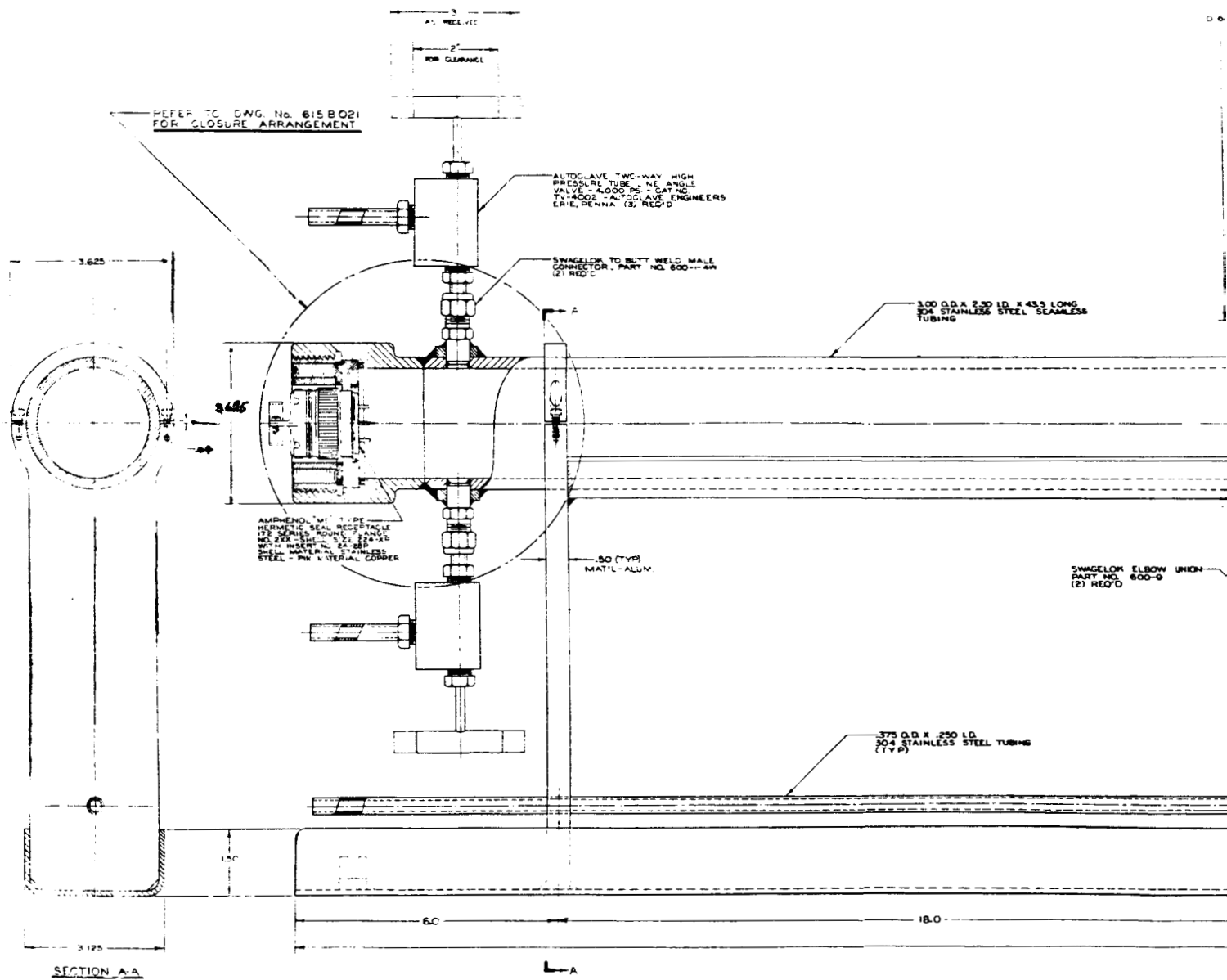
Fuel samples were instrumented with eleven to twelve 0.010 inch diameter tungsten-5% rhenium vs. tungsten-26% rhenium thermocouples. These were located at representative radial locations in the fuel matrix. The thermocouple wires inside the graphite fuel and inside the graphite crucibles were insulated with 0.030 inch outside diameter hard fired thoria. The lead wires inside the autoclave were insulated with fiberglass insulation. Leads terminated at a sealed pin receptacle, the flange of which was welded into the autoclave cover plate. The receptacle was mated by a plug attached to 70 feet of compensated lead wire cable. Three flux wires (1 per cent Co in aluminum) were attached axially, 120 degrees apart, along the outside of each crucible assembly.

C. Description of Fuel Material Tested

Type A - As manufactured 19 hole, NbC lined, beaded NERVA fuel material cut from fuel elements manufactured at Astrofuel Facility. The fuel samples were 6 inches long.

Type B - 7 hole, NbC lined beaded NERVA fuel material cut from the center of as-manufactured hexagonal fuel manufactured at Astrofuel Facility. Dimensions of fuel samples were 3/4 inches long, 1/2 inches diameter. Table 1 provides additional fuel sample data.

* Top and bottom is with reference to the autoclave removable closure which is referred to as the top of the autoclave, even though during tests the autoclave axis was horizontal.



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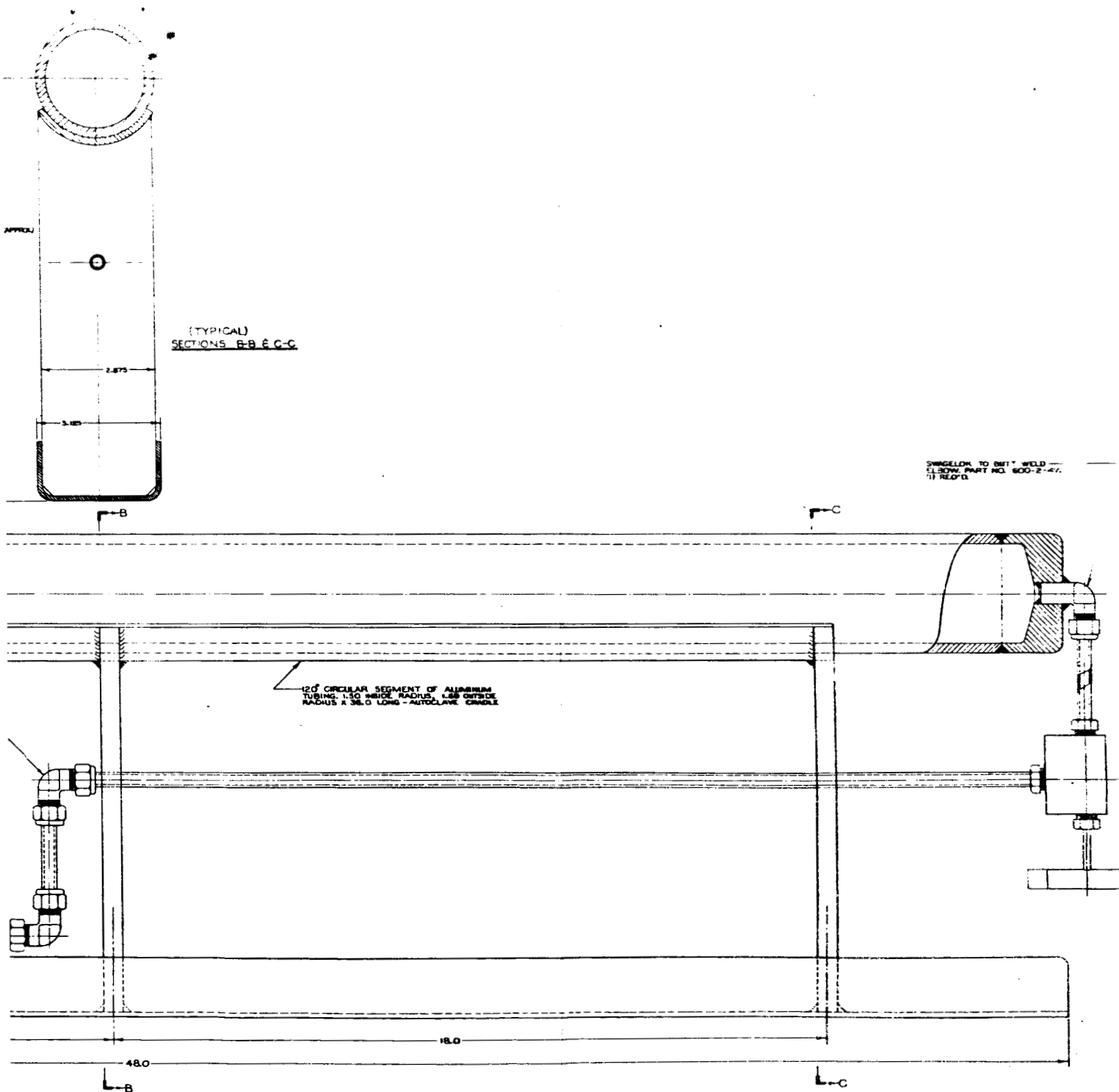


Figure 2 - Autoclave Capsule-TREAT Reactor

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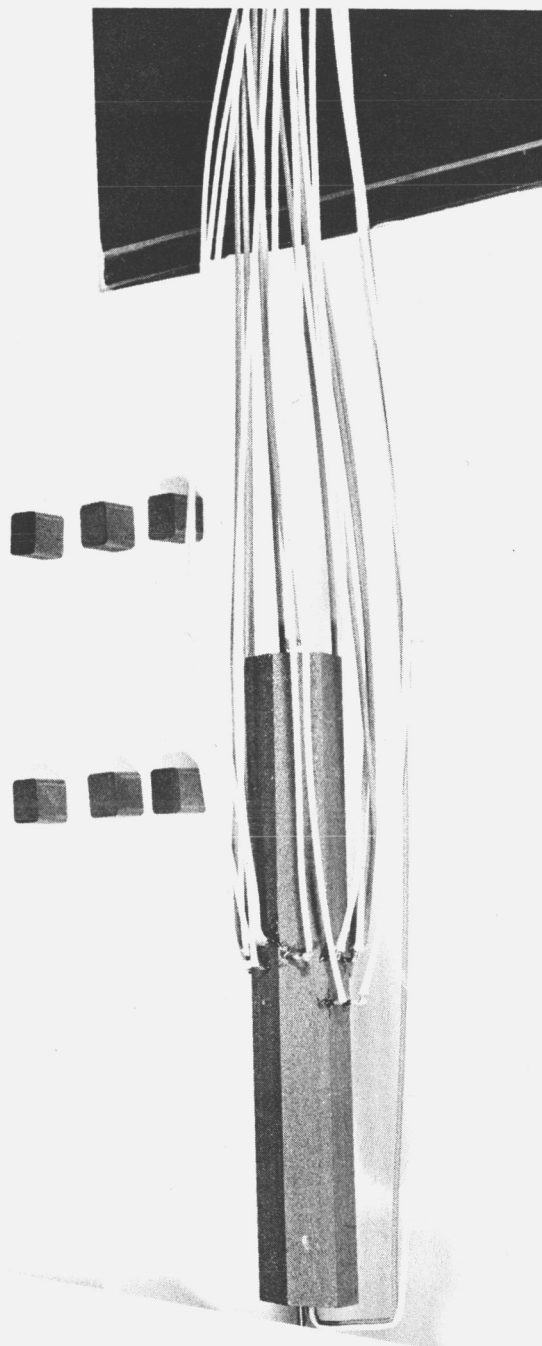
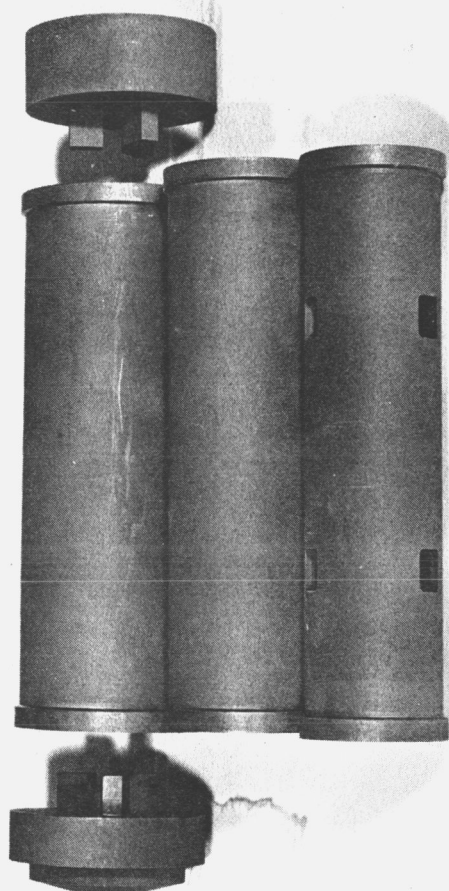


Figure 3 - Graphite Crucibles and Test Sample Assembly

Table I - Fuel Sample Data

<u>Exp. No.</u>	<u>Fuel Type</u>	<u>Fuel Element No.</u>	<u>Wt. of Sample (gm)</u>	<u>Enriched Uranium (%)</u>
A-1	A	99-04314	83.50	18.65
A-2	A	99-04459	82.46	18.26
	B	--	4.10	18.17
A-3	A	99-04459	79.85	18.26
	B	--	4.13	18.17
A-4	A	99-04468	79.91	18.26

IV. Results

The experimental parameters of the thermal stress experiments in TREAT are summarized in Table II. The calculated values of maximum sample temperature, maximum sample radial temperature difference, tangential tensile stress at the sample surface and the post-irradiation macroscopic and microscopic examination of the fuel material are summarized in Table III and in Table IV. The analytical methods which were used to arrive at the values in Table III and IV are given in the Appendix. Figure 4 shows the A-3 hexagonal fuel material before and after the transient. Figure 5 shows a macrophoto of the A-3 cylinder after the transient showing surface chipping or cracking.

V. Discussion of Results

Table III indicates clearly that no structural matrix damage occurred in the hexagonal samples tested during this series and that only a very slight surface chip resulted on one of the cylindrical samples tested during the current test series. The tensile strength of fueled graphite perpendicular to the axis of extrusion has been estimated to be about 5000 psi at 2500°C. The calculations indicate a maximum stress level of 1800 psi and thus predict that failure should not have occurred during these experiments.

One significant result is the difference in behavior between the cylindrical samples tested during these experiments and the cylindrical samples CEN-125, 126, 127, and 128 (LASL fuel) tested earlier at Argonne National Laboratory. * For reference the results of CEN-128, 129, and T-17 are listed in Table IV. The earlier work showed maximum cracking at a calculated stress level of 920 psi (CEN-128) and indicated a decreasing damage level with increasing temperature difference. A3-B which was subjected to a stress level of 1770 psi which was considerably above CEN-128 showed only very slight surface cracking. One conclusion which may be drawn is that the behavior of the LASL fuel previously tested was anomalous, possibly because it was laboratory prepared (pre-production) beaded KIWI fuel which was weaker than the WANL produced fuel material used for these experiments.

During the flat-top transients the fuel material remained at the test temperature for relatively long times. During these experiments heat loss occurred by radiation from

* Vogel, M. A., WANL-TNR-139, Irradiation Studies on NERVA Fuel Material - CY 1963.

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Table II - Summary of Experimental Parameters of Thermal Stress Experiments in TREAT

Exp. No.	Fuel Type Tested	Reactor Conditions			Fuel Conditions		
		Type of Pulse	Energy (mw-sec)	Period (ms)	Max. Power (mw)	Max. Measured Temp. (°C)	Time at Temp. (sec)
A1-1	A	Transient	94	109	402	774*	2-4
1-2	A	Transient	185	109	510	1733*	2-4
1-3	A	Transient	250	109	513	1930*****	2-4
A2	A	Transient	295	110	500	2200*****	2-4
	B	Transient	295	110	500	2400*****	2-4
A3	A	Transient	400	77	912	2760*****	2-4
	B	Transient	400	77	912	3010*****	2-4
A4	A	Flat-top	569	420	50	1855*	8
A5-1	A	Flat-top	567	623	48	2600**	9
5-2	A	Flat-top	531	680	22	2100**	17
5-3	A	Flat-top	236***	261	101	3140*****	2
							Power Density (Kw/cm ³)
							15.3
							19.4
							19.5
							19.0
							20.7
							34.7
							37.8
							1.90
							1.83
							0.84
							3.85

* Average of maximum temperature as indicated by nine thermocouples located at various radial positions on axial mid-plane of 6 inch long hexagonal sample.

** Maximum temperature as measured by one thermocouple in center hole of fuel material.

*** Reactor scrammed before flat-top was terminated.

**** Power density in fuel material during peak of pulse for transients; for 10-20 sec for flat-top pulses.

***** Estimated from previous experiments carried out at Argonne National Laboratory.

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Table III - Summary of Calculated Results and Post-irradiation Examination of Thermal Stress Experiments on WANL Fuel Material

Exp. No.	Fuel Type	Max. Fuel Temp. (°C)	Temperature Difference (between center and surface)		Tensile Stress at Surface (psi)	Post-test Visual Observations
			(°C)	(°C/inch)		
A1	A	1930*	175	605	770	No cracking of graphite matrix; NbC liner intact; porous surface due to ejection of UC ₂ beads
A2	A	2200	250	865	1060	"
A3	A	2760	440	1520	1800	No cracking of graphite matrix; NbC liner particles flaked off from graphite matrix; porous surface due to ejection of UC ₂ beads
A2	B	2400	235	940	1050	No cracking of graphite matrix; NbC liner intact; porous surface due to ejected UC ₂ beads
A3	B	3010	415	1660	1770	1/16-inch surface chip; NbC liner particles flaked off from graphite matrix, porous surface due to ejected UC ₂ beads
A4	A	2640	**	**	**	No cracking of graphite matrix, NbC liner intact, surface smooth.
A5	A	3140*	**	**	**	"

* Results of multiple transient tests A1 and A5 are for most energetic pulse.

** Temperature difference not calculated for A4 and A5 experiment because complex mode of heat transfer was not amenable to a simple model.

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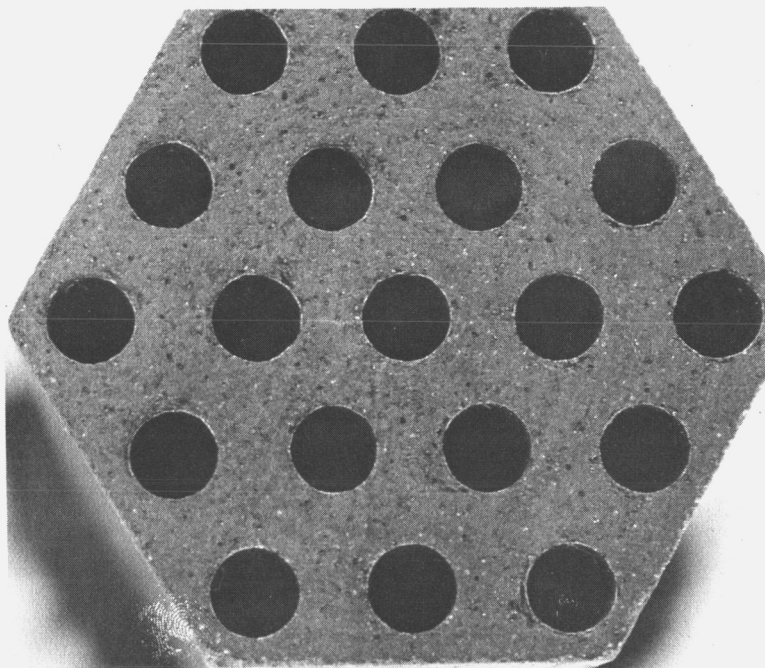
Table IV - Summary of Calculated Results and Post-irradiation Examination of Thermal Stress Experiments on Pre-production LASL (KIWI) Fuel Material

Experiment No.	Fuel Material Description	Max. Fuel Temp. °C	Maximum Radial Temperature Difference		Post-test Visual Observations
			(between center and surface) °C	Tensile Stress at Surface psi	
CEN-128	Type B not lined with NbC in channels	2275	200	800	920
CEN-129	"	2450	250	1000	1110
I-17	Type A 1 1/2 inches long	3350	680	2350	2850

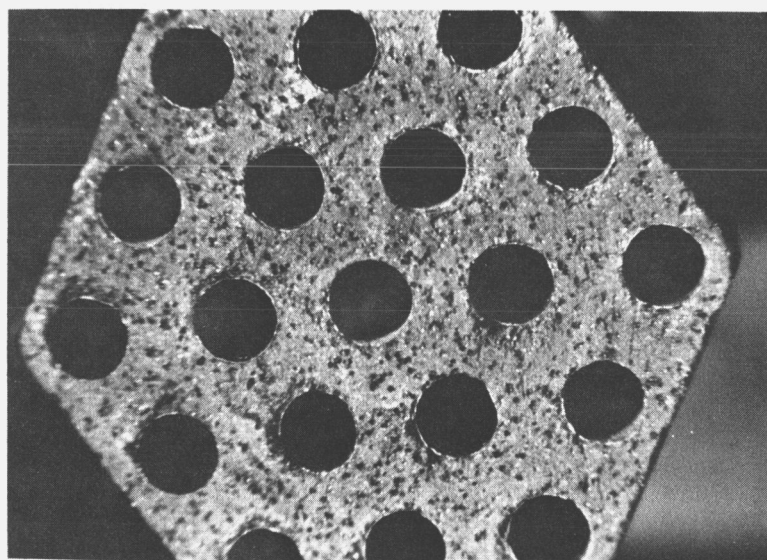
Fractured longitudinally into three segments; porous surface due to ejected UC₂ beads

1/4-inch longitudinal surface crack; porous surface due to ejected UC₂ beads

Radial crack near apex, porous surface and matrix due to bead ejection and graphite vaporization; NbC liner melted



Before



After

Figure 4 - A3 Fuel Material Before and After Test

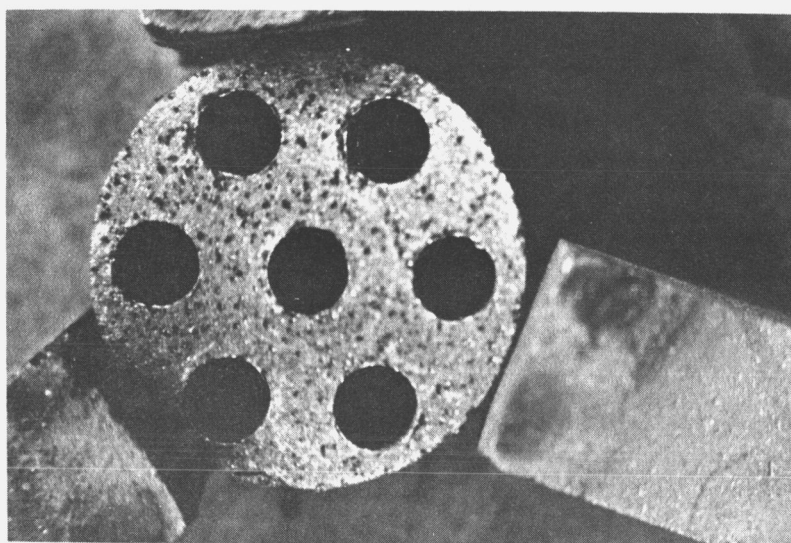


Figure 5 - A3 Cylindrical Fuel Material after
Test Showing Surface Chip

the fuel surface and by natural convection inside the coolant channels and on the fuel surface. The temperature differences resulting from this mode of heat transfer could not be calculated using the simple radiation heat transfer model which was used for the short duration pulses. However, it can be shown, at least qualitatively, that the flat-top pulses resulted in larger temperature differences than those which will exist in the NERVA core at the same power density. It is also significant that the A-5 sample was cycled three times from ambient to elevated test temperatures in excess of 2100°C without failure. Thus, the A-4 and A-5 tests confirm that the fuel material will be able to withstand the nominal NERVA core operating stresses.

VI. Conclusions

The work completed in TREAT indicates that as-manufactured (virgin) NERVA fuel material will suffer no damage from the normal operational steady-state or cyclic thermal stresses.

In addition, the fuel material can withstand thermal stress levels above 1800 psi produced by radial temperature differences up to 1520°C/inch which might result from plugged core channels, peripheral fuel element power peaking and/or cold hydrogen in-leakage.

VII. Acknowledgement

The author wishes to acknowledge the valuable assistance and contributions of the following:

Messrs. R. A. Fiore and R. L. Patterson at Westinghouse Astronuclear, who assisted in carrying out the experiments in Idaho and aided in performing the post-irradiation hot cell examinations.

The entire staff at the TREAT Reactor Facility under the supervision of Mr. J. F. Boland and Dr. H. Lawroski, who carried out the pulsed irradiations of the fuel material.

VIII. Appendix

This appendix outlines the methods used in the determination of temperature levels, temperature differences, and elastic thermal stresses reported in the body of the text.

A. Temperature Level

The experimental plans involved measurement of representative radial and axial fuel matrix temperatures during each pulse so as to give experimental values of temperature differences. During the initial calibration pulses (Capsule A-1), it became apparent that the bare wire thermocouples were not providing reliable readings above 1800°C. Post-irradiation examination revealed that the fired thoria insulation between the tungsten wires and the fuel material had reacted with the graphite even during the short time (2-4 seconds) involved in the pulses.

Fuel element temperature levels for experiment A1, A2, and A3 were calculated from previous calibrations of reactor energy levels versus fuel center temperature adjusted for the fission parameter* applicable to the fuel element-autoclave combination. The fission parameter was based on the average of three independent experimental methods; namely, a comparison of temperature measurements of fuel during the A1-1 and A1-2 pulse with previous temperature measurements at the same reactor energy (during which thermocouples were operative), radiochemical burnup analysis of A-4 fuel material (La^{140} and Zr^{95} nuclides) and activity level of the 1 per cent Co-Al wires in all four capsules (adjusted for flux depression in the fuel $\phi_{\text{avg}}/\phi_{\text{surface}} = 0.855$) Figure 6 shows the relationship between reactor energy and maximum Type A fuel sample temperature during a pulse. Type B fuel sample temperatures were determined by applying a correction factor of 1.09 (to account for a different flux depression) to Figure 6.

Fuel temperatures for the flat-top experiments A4 and A5 test were calculated from the previously determined fission parameter and a knowledge of heat loss as a

* Fission parameter - Number of fissions per Mw-sec of TREAT energy per gram of U-235. Average value used: 3.36×10^{12} fissions/(Mw-sec) (gm U-235).

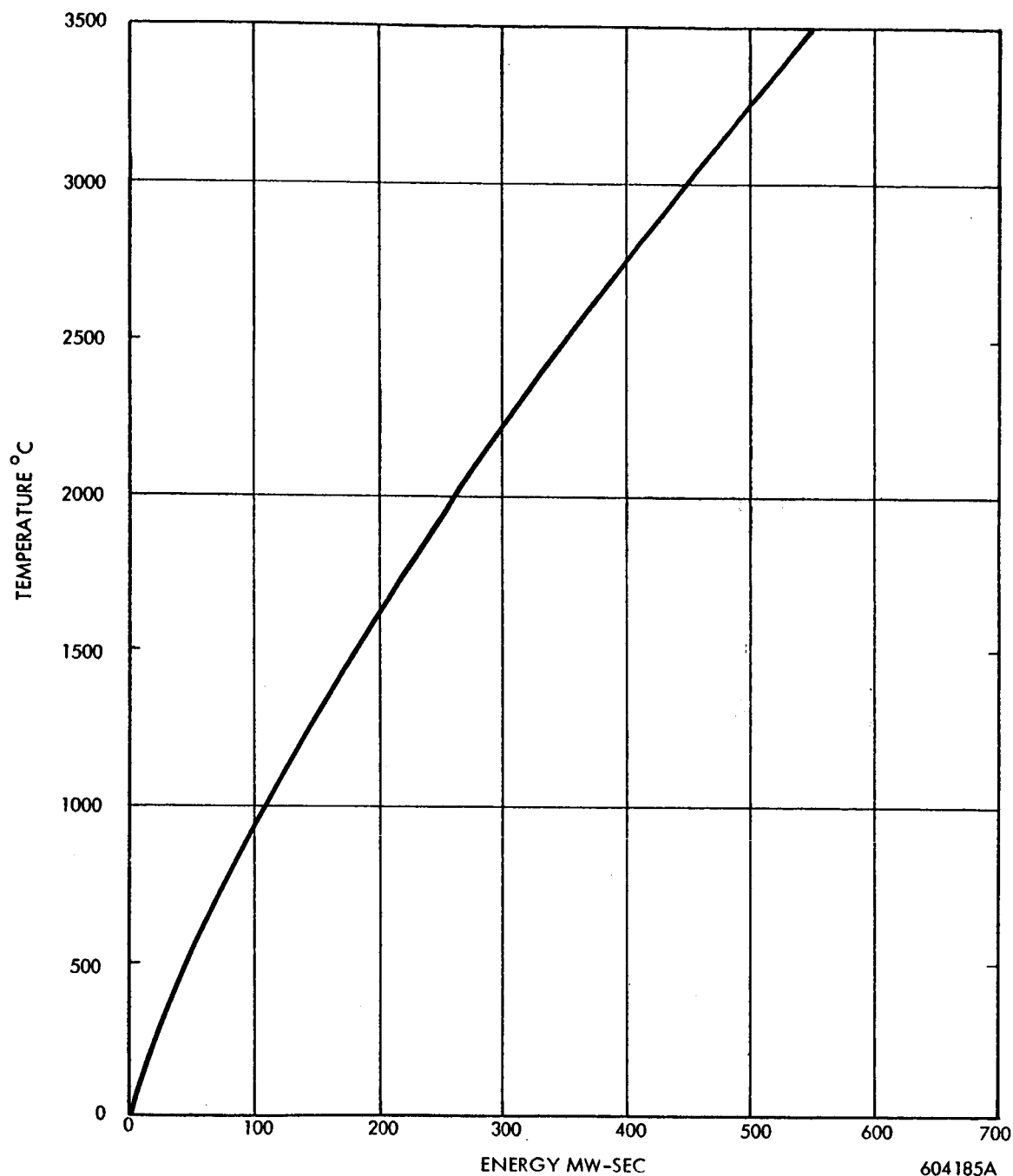


Figure 6 - Maximum Sample Temperature vs. Reactor Energy (TREAT)

$$F = 3.36 \times 10^{12} \text{ fissions/(gm U235) (mw-sec)}$$

17.0 w/o U235

function of sample temperature. These values were then adjusted on the basis of two measured temperatures during the A5 experiment. Figure 7 shows sample temperatures in terms of sample power density and reactor power for TREAT flat-top (extended duration pulses).

B. Maximum Radial Temperature Difference

Maximum radial temperature differences during a pulse were determined by transient heat conduction methods.* A number of simplifying assumptions were made in order to permit use of the available transient heat flow tables. Both types of samples tested were assumed to be solid cylinders infinitely long. The hexagonal shape (3/4 inch across flats) was turned into an equivalent cylinder of 0.787 inch diameter. Thermal properties of fuel material were assumed constant with temperature. (Values used were: heat capacity = 0.445 cal/(gm) (°C), density = 2.1 gm/cm³, thermal conductivity = 0.225 W/(cm) (°C). The fuel material heating and cooling cycle during each pulse was assumed to be as follows: adiabatic** nuclear heating to the maximum fuel temperature indicated in Figure 6 followed by thermal radiation cooling of the fuel sample surface. Heat was assumed to radiate to the inner graphite crucible at a rate proportional to the fourth powers of the temperature difference between the fuel surface temperature (at the time of maximum temperature difference) and the inside graphite crucible. The graphite crucible temperature was assumed constant with time (422°K) and for all fuel temperatures. The heat transfer coefficient (h) was assumed to be constant for all surface temperatures from time zero to the time of maximum temperature difference. It was found that this assumption greatly eases calculations and produces very little error in the results. Figure 8 plots maximum center to surface temperature difference as a function of maximum fuel temperature.

In experiments A4 and A5 the time at temperature (10-20 sec) was sufficiently long to initiate natural convection in the fuel channels in addition to thermal radiation from the fuel surface. These combined modes of heat transfer are not amenable to a calculation of temperature differences without a knowledge of pertinent boundary conditions.

* Austin, "Flow of Heat in Metals".

** Previous computer work showed that this is a reasonable assumption. WANL-TNR-139, op. cit.

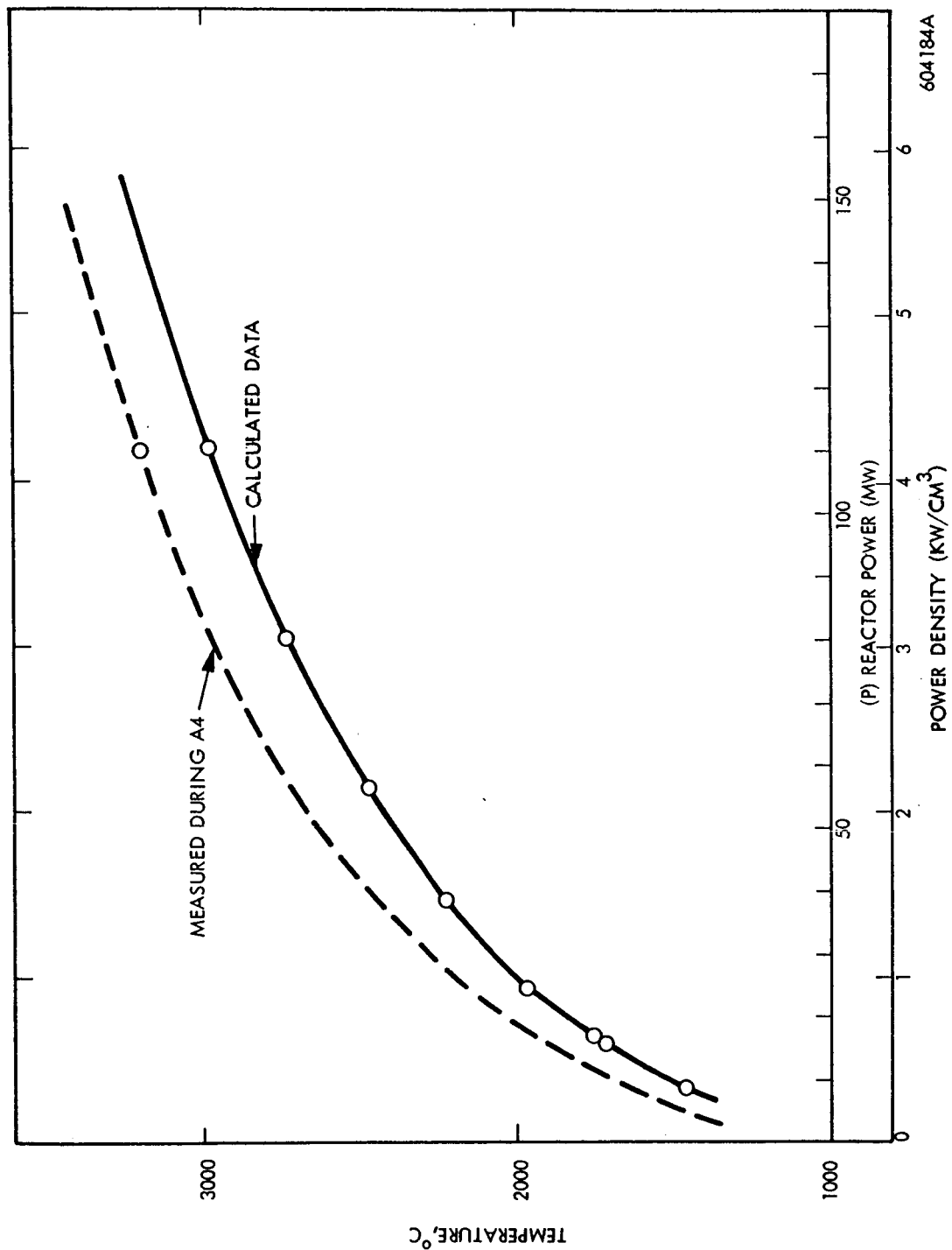


Figure 7 - Maximum Sample Temperature vs. Reactor Power and Sample Power Density

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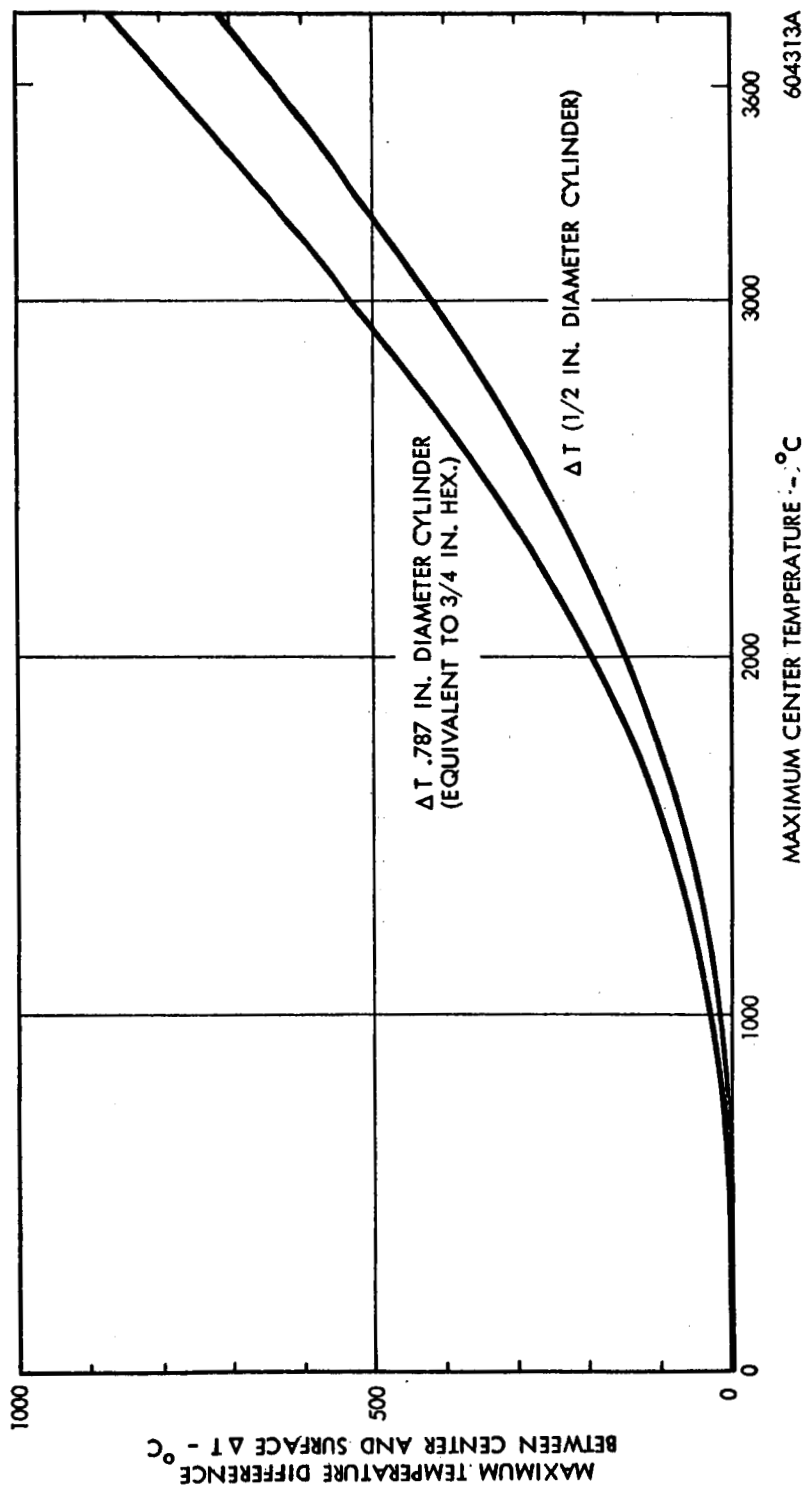


Figure 8 - Maximum Sample Temperature Difference vs. Maximum Sample Temperature

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Consequently, the radial temperature differences were not calculated for these experiments.

C. Thermal Stress at Sample Surface

Maximum elastic tangential thermal stresses at the cylinder surfaces were calculated for the pulsed experiments A1, A2, and A3 using the previously determined transient conduction constants in transient thermal stress equations developed by J. C. Jaeger.* It follows that these numerical results involved the same assumptions and the same cooling mechanism as previously described. In addition, the thermal stress equations apply strictly only to isotropic materials without holes. However, directional material properties (perpendicular to extrusion axis) were used in calculating the stresses. The property values used were at a temperature of 2000°C and are based on recent measurements at WANL. These are coefficient of thermal expansion (α) = $3.65 \times 10^{-6}/^{\circ}\text{C}$ modulus of elasticity (E) = 2.0×10^6 psi and Poisson's ratio = 0.15.

Table V summarizes the calculated data of radial temperature differences and tangential surface stresses which forms the basis of Figure 8 and Figure 9.

* J. C. Jaeger, "On Thermal Stresses in Circular Cylinders," Philosophical Magazine, Vol. 36, 1945, p. 418.

Table V - Calculated Radial Temperature Differences and Tangential Thermal Stresses as a Function of Maximum Sample Temperature

Maximum Sample Temperature		Maximum Sample Temperature Difference Between Center and Surface, °C		Maximum Elastic Tangential Thermal Stress on Surface, psi	
Maximum Sample Temperature, °C		(1/2 inch cylinder)	(3/4 inch hex)	(1/2 inch cylinder)	(3/4 inch hex)
1510	1201	61	47	350	271
2102	1752	162	133	755	620
3155	2607	468	382	1945	1590
4068*	3366	836	685	3540	2910
4964*	4251*	1254	1068	5380	4580

* These temperatures exceed the sublimation temperature of graphite $\approx 3600^{\circ}\text{C}$ and could not be attained in practice; however, they show theoretically the effect of higher initial sample temperature on stress level attained during cooldown.

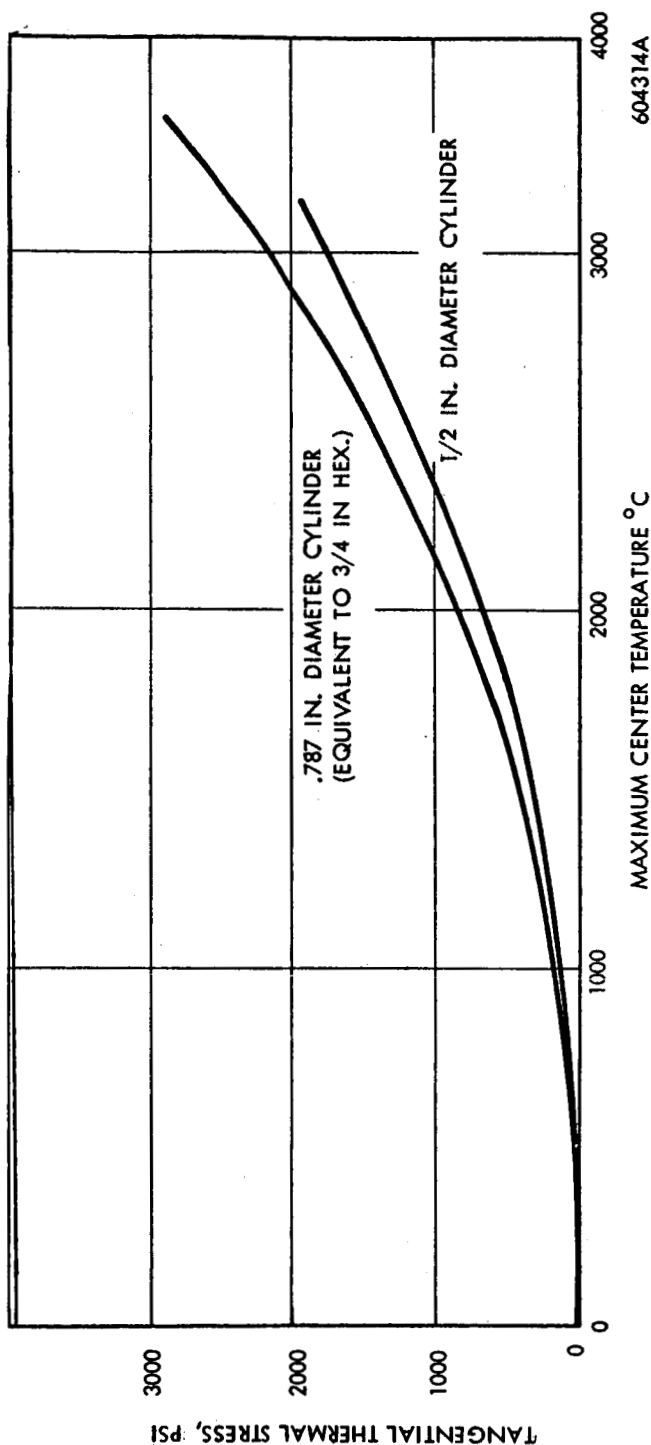


Figure 9 - Maximum Elastic Tangential Thermal Stress at Surface vs. Maximum Sample Temperature